

NON-PUBLIC?: N
ACCESSION #: 9510020133
LICENSEE EVENT REPORT (LER)

FACILITY NAME: St. Lucie Unit 1 PAGE: 1 OF 5

DOCKET NUMBER: 05000335

TITLE: Automatic Reactor Trip During Turbine Overspeed
Surveillance Testing due to Personnel Error.
EVENT DATE: 07/08/95 LER #: 95-003-1 REPORT DATE: 09/23/95

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Edwin J. Benken, Licensing Engineer TELEPHONE: (407) 468-4248

COMPONENT FAILURE DESCRIPTION:
CAUSE: A SYSTEM: AB COMPONENT: RV MANUFACTURER: D243
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On July 8, 1995, Unit 1 was operating at 100 percent reactor power. Operations personnel were conducting a scheduled Turbine overspeed trip surveillance per an approved plant procedure. During the portion of the surveillance that tests a solenoid valve for Overspeed Protection Control (20-1 OPC) a utility non-licensed operator failed to close an isolation valve as directed by the procedure. Failure to close this valve allowed electro-hydraulic (EH) fluid from the Governor valves (GV) and Intercept valves (IV) to drain when the solenoid valve was opened in a subsequent step. Draining of the EH fluid caused closure of the Main Turbine Governor and Intercept valves which resulted in an automatic reactor trip.

The root cause of this event was cognitive personnel error on the part of a utility non-licensed operator who failed to properly implement a procedural step during performance of a surveillance.

Corrective actions for this event: 1) Operations personnel involved with the event were counselled. 2) Procedure changes are being made to incorporate human factors improvements and additional step verifications. 3) Other load threatening surveillances are being reviewed to determine if generic change are warranted. 4) A technical subcommittee is evaluating this event for additional corrective actions to prevent reoccurrence. 5) Site management held a trip review meeting open to all disciplines for lessons learned from this event.

END OF ABSTRACT

TEXT PAGE 2 OF 5

DESCRIPTION OF THE EVENT

On July 8, 1995, St. Lucie Unit 1 was operating at 100 percent Reactor power. A utility non-licensed Operator was performing the monthly turbine overspeed trip test in accordance with an approved plant procedure. The non-licensed operator was performing the steps of the procedure while a utility licensed Operator maintained radio communication with the control room.

During the portion of the test which checks the operability of an Overspeed Protection Control (OPC) solenoid valve, SE22138 (EHS:TG), the procedure directed the operator to unlock and close V22482 (EHS:TG), "EH Test Header to 20-1/OPC Isolation." This is the electro-hydraulic (EH) fluid inlet isolation to the OPC solenoid valve. This step ensures that the OPC solenoid valve is isolated from the actual EH fluid system (EHS:TG) supplying the turbine Governor (GV) and Intercept valves (IV) (EHS:SB) prior to testing the solenoid. The NPO removed the locking device from isolation valve V22482, but was momentarily distracted by placing the locking device in a secure position, and failed to close the valve as directed by the procedure. When the next step of the procedure was executed (the actual stroke testing of solenoid valve SE22138) EH fluid was drained from the GVs and IVs causing the GVs and IVs to rapidly close. Closure of the turbine valves quickly reduced steam flow through the turbine which resulted in a reactor trip from high pressurizer pressure at 1122 hours. The maximum RCS pressure reached during this event was approximately 2430 psia. The maximum secondary pressure reached was approximately 1023 psia.

Emergency Operating Procedure (EOP)-1, "Standard Post Trip Actions" was immediately implemented. Operators observed increasing level in the 1A SG after the trip and closed the 15 percent feedwater bypass valve. Level continued to increase and the Control Room Operators closed the

isolation valve for the 1A Feedwater Regulating Valve (EIIS:JB). The 1B Main Feedwater Pump (MFW)(EIIS:SJ) subsequently tripped from a low flow condition, and the 1A MFW Pump tripped due to high level in the 1A SG. The 1B MFW Pump was restarted and SG levels were then controlled within the normal band.

A relief valve in the Letdown Level Control System (EIIS:CB) opened during the event due to the system transient, and subsequently closed when Control Room operators reduced the letdown pressure controller (EIIS:CB) setpoint. The Steam Generator Safety Valves (EIIS:SB) functioned as designed to limit SG pressure during the initial transient. The Steam Bypass Control System (SBCS) (EIIS:JI) functioned properly to control RCS temperature during this event.

The Control Room crew completed the actions of EOP-01, "Standard Post Trip Actions", and implemented EOP-02, "Reactor Trip Recovery" after diagnosing an uncomplicated trip. Upon completion of the Reactor Trip Recovery procedure, the unit was maintained in a stable, Mode 3 condition for post trip review and event investigation.

TEXT PAGE 3 OF 5

CAUSE OF THE EVENT

The cause of this event was cognitive personnel error by a utility non-licensed operator who failed to correctly implement a procedural step during performance of a turbine overspeed trip surveillance. The operator was momentarily distracted by placing a valve locking device in a secure position, and did not close the valve as directed by the procedure.

ANALYSIS OF THE EVENT

This event is reportable under the requirements of 10 CFR 50.73.a. 2.iv, as "any event that resulted in a manual or automatic action of any Engineered Safety Feature."

The closure of the Main Turbine Governor and Intercept valves caused a rapid reduction in secondary steam flow. The effect of the reduction in secondary steam demand was an increase in SG pressure and temperature, and RCS temperature and pressure. Increasing RCS pressure resulted in an uncomplicated Reactor trip on high pressurizer pressure as designed.

An investigation performed after the event revealed that the calibration on the 1A Main Feedwater Regulating Valve (FCV-9011) electro-pneumatic transducer (E/P) had drifted, so that the feedwater flow control valve

did not close fully as expected on the plant trip. This caused the 1 A Steam Generator level to increase above the normal value to the high level trip setpoint for the Main Feedwater Pump. Closing the Main Feedwater Block valve secured the flow to the 1A SG from FCV-9011, stabilizing SG level.

This event is bounded by section 15.2.7 of the St. Lucie Unit 1 Updated Final Safety Analysis Report (UFSAR) "Loss of External Electrical Load or Turbine Stop Valve Closure." This section describes a rapid, large reduction of power demand on the reactor while operating at full power. The UFSAR states, "When the turbine stop/control valve closes, the steam flow is terminated, causing the secondary system temperature and pressure to increase. The primary-to-secondary heat transfer decreases as secondary system temperature increases. If the reactor is not tripped when the turbine is tripped the reactor will trip on high pressurizer pressure, reducing the primary heat source."

In addition to the above, UFSAR section 15.2.7, states that, "The mitigative features of the pressurizer spray, pressurizer relief valves (PORV), and the Steam Bypass System are assumed not to function so as to exacerbate the calculated pressurization of the primary system. The purpose is to demonstrate that the primary safety relief capability is sufficient to limit primary pressure to less than 110% of the design pressure (2750 psia), and to demonstrate that the secondary safety relief capacity is sufficient to limit secondary pressure to less than 110% of the design pressure (1100 psia)."

TEXT PAGE 4 OF 5

ANALYSIS OF THE EVENT (continued)

During this event, the maximum primary pressure reached was approximately 2430 psia, which is below the Pressurizer code safety valve setpoint of 2500 psia. During the initial review of plant data from the reactor trip, it was determined that the PORVs (EIIS:AB) functioned properly to limit primary pressure. This conclusion was based on a review of the primary pressure response and PORV acoustic flow data, and supported by a Quench Tank pressure increase seen during the transient. In August of 1995, the PORV main valves were found to be inoperable due to improper mechanical assembly (Reference LER 335-95-005-00). Subsequent testing, inspection and analysis showed that the PORV main valves most probably did not open during this event. The (SG) code safeties (EIIS:SB) operated to limit SG pressure to 1023 psia and the SBCS functioned as designed. These systems, in conjunction with the Reactor trip, functioned to limit primary system pressure. This event is less limiting than that described in UFSAR section 1 5.2.7. The health and safety of

the public were not affected by this event.

CORRECTIVE ACTIONS

- 1) Operations personnel involved with this event were counseled on the importance of applying self-checking principles.
- 2) The surveillance procedure for conducting this test, OP 1/2-00301 50, "Secondary Plant Operating Checks and Tests" will be changed to incorporate format improvements, and to include additional verification that critical steps have been completed.
- 3) Plant Staff will review other load threatening surveillances to determine if additional procedural changes or precautions are necessary to minimize the potential for personnel error.
- 4) A technical subcommittee was formed to evaluate this event for generic implications and provide additional corrective actions to prevent reoccurrence.
- 5) Site management held a trip review meeting, attended by personnel from Operations, Maintenance, Training, Engineering, Technical staff, and senior Nuclear Division management to examine this event. The meeting was video taped to assure that lessons learned are available to all Operations personnel.
- 6) Instrument and Control (I/C) and System Engineers calibrated the 1A Main Feedwater Regulating Valve E/P transducer prior to unit startup. The Main Feedwater Regulating valve positioning components affecting this event are being evaluated for additional corrective actions.
- 7) This Event will be included into Operations training for both licensed and non-licensed Operations personnel.

TEXT PAGE 5 OF 5

ADDITIONAL INFORMATION

Failed Component Identification

Manufacturer: Dresser Ind. VIv & Inst Div / Ashcroft
Model Number: 31533VX-30
Device: Pressurizer PORV valve

Previous Similar Events

LER 389/86-002 describes a Reactor trip initiated by loss of load during Turbine overspeed testing due to cognitive personnel error.

ATTACHMENT TO 9510020133 PAGE 1 OF 1

Florida Power & Light Company, P.O. Box 128,
Fort Pierce, FL 34954-0128

September 23, 1995

FPL

L-95-269
10 CFR 50.73

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Re: St. Lucie Unit 1
Docket No. 50-335
Reportable Event: 95-003 - Revision 1
Date of Event: July 8, 1995
Automatic Reactor Trip During Turbine Overspeed surveillance Testing
due to Personnel Error

The attached Licensee Event Report is being submitted pursuant to the requirements of 10 CFR 50.73 to provide an update on the subject event.

Very truly yours,

D. A Sager
Vice President
St. Lucie Plant

DAS/EJB

Attachment

cc: Stewart D. Ebnetter, Regional Administrator, USNRC Region II
Senior Resident Inspector, USNRC, St. Lucie Plant

an FPL Group company

*** END OF DOCUMENT ***
